Nuclear Safety Oversight Committee

REPORT OF

THE REACTOR SAFETY RESEARCH REVIEW GROUP

Norman C. Rasmussen, Chairman
Herbert J. C. Kouts, Vice Chairman

Spencer H. Bush
Thomas J. Connolly
Herbert G. MacPherson
David Okrent
Lombard Squires
Edwin L. Zebroski

SUBMITTED TO

THE PRESIDENT'S NUCLEAR SAFETY OVERSIGHT COMMITTEE

Governor Bruce Babbitt, Chairman

John Deutch
Marvin Goldberger
Harold Lewis

Steven Ebbin, Executive Director

SEPTEMBER 1981

September 1981

The Honorable Bruce Babbitt
Chairman
Nuclear Safety Oversight Committee
Washington, D.C. 20545

Dear Gov. Babbitt:

I am pleased to transmit herewith the report of the Reactor Safety Research Review Group. You will note that it contains a number of recommendations directed toward improving the national reactor safety research program, with particular attention to the Nuclear Regulatory Commission research program.

The text contains a number of recommendations considered important enough to be singled out. Of these, 26 were considered to be of a higher level of urgency and these have been listed at the front of the report for emphasis.

We hope these recommendations will be of use to your committee in carrying out its responsibilities. I would like to take this opportunity to thank you for your support and that of your staff, especially Mr. Steven Ebbin, in helping us carry out this review.

I look forward to appearing before your committee on September 23, 1981 to discuss this report in person.

Very truly yours,

Norman C. Rasmussen

Norman C. Rasmussen

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Introduction

Scope of Review

This review of the national program of reactor safety research was performed at the request of the Nuclear Safety Oversight Committee. It has been limited to research related to safety of nuclear power plants, and specifically does not include other aspects of the nuclear fuel cycle. It has been further restricted to safety of light water moderated and cooled nuclear power plants of the types used in the United States.

The Review Group

To conduct the review, NSOC selected a group of individuals with extensive experience in reactor safety issues. Professor Norman C. Rasmussen of the Department of Nuclear Engineering at the Massachusetts Institute of Technology was asked to chair this Review Group. The other members were:

Dr. Spencer Bush	Battelle Pacific Northwest Laboratories
Dr. Thomas Connolly	Stanford University
Dr. Herbert J.C. Kouts	Brookhaven National Laboratory
Dr. Herbert G. MacPherson	Institute for Energy Analysis
Dr. David Okrent	University of California at Los Angeles
Mr. Lombard Squires (retired)	E.I. du Pont de Nemours Company
Dr. Edwin Zebroski	Nuclear Safety Analysis Center

Performance of the Review

In performing its review, the Group had the benefit of meetings with a number of members of the U.S. water reactor safety research community. An organizational meeting was held on April 24, 1981, in Chicago. A second meeting was held on June 4-5 in Washington,D.C., with representatives of NRC's Office of Nuclear Regulatory Research and representatives of DOE. On June 25-26, a meeting was held in Denver with representatives of the four LWR vendors (Westinghouse, Combustion Engineering, Babcock and Wilcox, and General Electric), selected utilities (Commonwealth Edison, Tennessee Valley Authority, Duke Power, and Pacific Gas and Electric), and individuals representing consulting companies (Saul Levine of NUS, Robert Budnitz of Teknekron, and Mario Fontana representing IDCOR*). In addition, individual members

* Industry Degraded Core Rulemaking Program

of the Review Group visited all of the National Laboratories where the bulk of the nation's safety research is done. Discussions were held there with staff members at both the managerial and working level. Comments and views expressed to the Review Group at both individual and group meetings have been factored into this report, but without attribution.

An extended meeting of the Review Group was held in Seattle from August 10-14, 1981. A final meeting was held in Washington on September 2, 1981.

Research Considered in the Review

Research on light water reactor safety is sponsored by many organizations. These include the Nuclear Regulatory Commission (NRC) and the Department of Energy (DOE) in the public sector, and reactor vendors and the utilities through owners' groups and the Electric Power Research Institute (EPRI) in the private sector. There is also a substantial and growing amount of water reactor safety research abroad, principally in the Federal Republic of Germany, France, Japan, and Sweden. Because many projects abroad are sponsored jointly with organizations in the U.S., and because the information developed in most national programs is fully exchanged, water reactor safety research should be viewed as an international effort. Some attention is therefore given here to contributions by other countries.

At the present time, more than half of the research on water reactor safety in the United States is sponsored by the NRC. In FY-82, for example, the budget of the Office of Nuclear Regulatory Research of NRC is $230 million, of which approximately $200 million is destined for light water reactor safety research. The research by DOE on this subject is very limited. Public Law 96-567 now calls for a substantial DOE role in this area. Implementation of this law is still under review, and budgetary allocations for it have yet to be made.

In the private sector, water reactor safety research is frequently sponsored by groups of utilities and/or nuclear steam supply vendors with specific common problems. The Electric Power Research Institute (EPRI), the research arm of the electric utilities, is also a sponsor of substantial research on light water reactor safety and reliability. In FY-82, EPRI's expenditures in this area are expected to amount to about $65 million. Since the accident at Three Mile Island, the industry has also formed the Nuclear Safety Analysis Center (NSAC) and the Institute for Nuclear Power Operations (INPO). These organizations sponsor additional activities to improve safety of light water reactors.

These estimates of $200 million expended annually in the public sector on light water reactor safety research, and perhaps half that value sponsored by the private sector, are necessarily arbitrary. Typical questions that arise in considering whether a given project by industry should be included are: is it research or is it simply engineering analysis, and is it research to improve reactor safety or to improve a vendor's product? For example, the question of whether a given research project is or is not safety research arises in connection with certain programs sponsored by owners' groups, which deal with product reliability (e.g., improvement of design of steam generators for PWRs). It is acceptable to argue that increased reliability leads to increased safety, but these programs are driven more by economic considerations than by safety considerations.

The NRC, on the other hand, also conducts many activities resembling research in its Technical Assistance Programs. These are not usually defined as research, even though some programs under the rubric of technical assistance have been long-term and have been exploratory in nature. These technical assistance programs have not been reviewed here.

Structure of the Report

The review that follows has been structured along the lines of accident analysis, because this brings up the safety issues in a natural way. The first section discusses research that helps to avoid accidents and research on the modes of accident initiation. This is followed by a section on research pertinent to the response of the plant to initiating events, including functioning of safety systems. The third includes research on the consequences within the plant if the protective systems were to fail; this includes fuel failure and liberation of fission products from fuel, transport of the fission products, and their release from the containment. The fourth section includes research on consequences of fission product release. The fifth includes questions which pervade the entire accident analysis. The sixth contains observations on organizational and institutional subjects.

Interspersed in the text of the report are a number of recommendations that the Review Group feels would improve the overall effectiveness of the national program. The most important of these are singled out in the text. Forty-two recommendations are emphasized in this way.

Conclusions and Recommendations

General Conclusion

The national research program on safety of light water reactors began to assume its present form and objectives only about eight years ago. The structure it assumed then was a reaction to questions raised during the long hearing of 1972-73 on criteria for effectiveness of emergency core cooling systems. The program has kept this initial orientation ever since, with little redirection to topics not originally included. This is understandable in the context of the length of time that has been found necessary to mount such a complex research program and to carry it to useful conclusions. In the intervening period, the program has made significant progress in answering questions on reactor safety. The NRC, its contractors, and those in industry who have joined in this program have earned and well deserve a substantial measure of credit for important contributions to the safety of the public.

The Large LOCA Program

In particular, the research to improve understanding of the large loss-of-coolant accident has been an unquestioned success. It has progressed to the point where the strong emphasis on this research objective can now be wound down. The LOFT program, which is the centerpiece of large LOCA research, has succeeded in its original objectives, and the Review Group agrees with views expressed elsewhere that moves to decommission LOFT can now be taken. In subsequent sections of this report, we identify high priority areas in which research results are urgently needed, and to which the resources saved through the phasing out of LOFT should be applied. Our recommendation is:

> The LOFT program should be phased out in an orderly manner. This Review Group is divided on whether the experimental program for LOFT should be completed in FY-82 or FY-83. (P.II-2)*

New Research

Very important issues have arisen in other areas in the past few years, and a number of these now compete for attention. Research programs have been started on some of these issues. In some other instances, though, research programs are required but do not exist. We have singled out a number of issues that urgently need

* Each recommendation repeated in this section is accompanied by the page number of the main report where it may be found.

research but on which no significant research is being done. Recommendations on these are given in the text of our report. The most important are listed below, in order of urgency and importance.

A major effort should be undertaken to develop and evaluate improved or alternate approaches to more reliable shutdown heat removal systems, for both the reactor vessel and the containment. (p. II-3)

A program should be initiated to develop and evaluate methods of providing more reliable electric station power, both AC and DC. Such a program would have significant value and an excellent cost-benefit ratio. (p. I-7)

A program should be undertaken to examine the relative safety merits of rigid versus flexible seismic design of piping. The program should consider allowance for inelastic behavior of piping and evaluate the potentially adverse impact of an excessive number of hangers and snubbers on in-service inspection and the possibility of crack initiation during normal operation due to snubber malfunction. (p. I-5)

Industry or DOE programs should be established to develop valves with more reliability of function under normal and abnormal operating conditions. (p. I-4)

DOE and industry should now proceed with the long-planned, comprehensive program to develop and demonstrate primary system decontamination techniques applicable to LWRs. This program should include techniques which reduce the initial deposition of radioactive materials. (p. I-5)

A national LWR system simulation program should be undertaken cooperatively by DOE and NRC. This program should treat both PWRs and BWRs generically. The principal goal should be the development of computational capability to study LWR behavior in real time or faster through a wide range of severe transients including accidents involving extensive core damage. (p. II-4)

NRC programs should be established to determine the mode and likelihood of severe damage by water hammer, to permit engineering solution to prevention of damage to the plant and especially to safety-related systems. (p. I-4)

The NRC should undertake the studies needed to develop design measures and/or other measures which protect against sabotage by an insider while not compromising safety in other respects. (p. I-7)

Existing Research

Among the research programs now being supported, several have received specific attention in our report. In most cases, this is because either increases or decreases in emphasis are appropriate. In a few cases we have said that current lev-

els of support are reasonable. Among our recommendations on existing research programs, the following are especially important. They are given in order of urgency and importance:*

> Probabilistic risk analysis indicates that about half the risk from reactor accidents is attributable to human error. The body of knowledge concerning pertinent human factors is inadequate, and it is important that further research be done in this area to provide an adequate technical basis for regulatory activities. The NRC should establish relative priorities, so that its research program will be structured to obtain the most important information first. The important information is that which can be used to improve control room design, operator aids, and selection and training of operators, which should lead to reduction of rates of human error. Further, the NRC program should be coordinated with industry efforts. (p. V-2)

> The NRC, DOE, and industry have ongoing programs to provide improved understanding of radioactive releases possible after severe reactor accidents. These should encompass a range of conditions and accident scenarios sufficient for broad understanding of the processes of transport and release of the important radioactive isotopes. The NRC program should provide as much information as possible for the degraded core cooling hearings. There is also a need for a long-range program in this area. (p. III-7)

> The NRC program on the small loss-of-coolant accident should be continued, to the point where capability to answer important questions of thermal-hydraulics and fuel performance is assured. (p. II-3)

> The possibility of cold repressurization of reactor pressure vessels that have undergone large shifts in nil ductility transition temperature provides a potential mechanism for a major failure of the pressure boundary. The substantial programs underway or in the advanced planning stage should be funded and coordinated to assure that the issue is resolved in a 1-3 year time frame. (p. I-3)

> Ample federal funds should be made available for developing timely information through examining the degraded core in TMI-2. The current program of DOE to carry out such investigations should be supported. (p. III-4)

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* Dr. Okrent has the following additional comment:
All of the above recommendations are important. If required to assign priorities, I would arrive at a somewhat different order. I would place the 4th, 9th, and 10th recommendations on cold repressurization, minimum engineered safety features, and the degraded core cooling rulemaking, respectively, in a group at the top of the list. I would place the 1st, 2nd, and 11th on human factors, the radioactive source term, and mitigation features, respectively, in the next highest group. I would place the 3rd, 7th, and 8th on the small LOCA, probabilistic analysis, and control systems, respectively, in a third group, while the 5th and 6th on the TMI-2 core and hydrogen, though still important, would be in a last group.

The NRC is initiating a research program on the issue of evolution and burning of hydrogen in accidents. This program is important and should be pursued expeditiously. Complementary programs by industry should be taken into account in structuring the NRC's research program. The significance of containment design and volume should also be taken into account. (p. III-9)

The use of probabilistic analysis is rapidly expanding. A significantly larger research program is needed to improve and standardize the methodology as much as possible. Particular areas that need improvement are: methods of handling common cause failures, data collection and analysis, accuracy of the estimated frequency of very low probability events, and assurance of the quality of analyses. (p. V-4)

Research on the role and importance of plant control systems in LWR safety should be greatly augmented. (p. I-7)

The NRC should promptly identify the information needs of the rulemaking on minimum engineered safety features of future LWRs that can be supplied in time by safety research, and assign the necessary priority and resources to this task. (p. II-5)

The NRC should promptly identify that information which it needs for the degraded core cooling rulemaking and which safety research can supply in time for the hearing. It should assign the priority and resources to get the job done, deferring, if necessary, other less urgent degraded core safety research. (p. III-4)

The NRC safety research program on mitigation of degraded core and core melt accidents should be modified as necessary to provide the information needed on alternative containment design concepts under consideration, including improved containment cooling, containment venting, venting and filtering, core debris retention, and hydrogen control. (p. III-10)

General Comments on NRC Research Programs

We also reviewed such pervasive matters as the logic and structure of NRC's safety research program, the way the benefits of research are incorporated (or not incorporated) into regulations, and the general relationship between the Office of Nuclear Regulatory Research and the other components of NRC. Among the observations we transmit are several recommendations on improvements to the way NRC plans, manages, and uses its research programs. The most important are:

The Commission should encourage and promote the visible integration of the results of safety research into the regulatory process. Regulatory requirements should not rest on bad science when good science has become available. If retention of conservatism is desired because of uncertainties or for other reasons, this should be done through the application of explicit safety factors to a calculation based on best available methods, in accordance with good engineering practice. (p. VI-3)

> The limitation of NRC's programs to "confirmatory research" should be removed, so that exploratory research and research to improve safety can be undertaken when this looks like the better course to follow. (p. VI-3)
>
> NRC's Long Range Research Plan (NUREG-0740) should be restructured to follow lines of Agency objectives. This should infuse it with a more up-to-date logical structure and discourage the tendency to support more of what has already been done. Introduction and continuation of guidance at the Commission level will be of fundamental importance in confirming the objectives and restructuring the plan. The restructured long-range plan should be more specific in defining the deliverables in each program. (p. VI-3)
>
> A policy should be instituted whereby probabilistic risk assessment is used as one tool in establishing priorities of research programs. (p. VI-6)

The last of the above recommendations should be read in a context broader than NRC programs on safety research. It is good advice for all applied research.

Research by DOE

In Public Law 96-567, Congress directed the Department of Energy to undertake studies leading to a program to contribute to sound design and safety of nuclear power plants. We have reviewed the DOE's reactions to these instructions and find them to be weak and generally unresponsive. We have two important recommendations on this subject, listed below:

> DOE should form a strong staff of technical and managerial personnel knowledgeable and experienced in the subject of water reactor safety, to develop and implement programs of safety research in this field. (p. VI-5)
>
> DOE should develop a program of generic research to improve water reactor safety, and assume a substantial responsibility in the area of accident prevention to supplement other programs designed to reduce the likelihood of nuclear power plant accidents. (p. VI-5)

I. Potential Accident Initiators and Their Prevention

For many years it has been well known that the occurrence of certain incidents during the operation of a nuclear power plant could start a chain of events leading to fuel damage and accidental release of radioactivity, if not responded to correctly. It is convenient to classify such events into those having external causes and those resulting from failures inside the plant. Risk analyses generally conclude that among the external causes, large earthquakes are the most likely to cause a serious accident. Other causes considered in safety analyses are floods, wind storms, external fires or explosions, and sometimes airplane crashes. Among the internal events considered are failures in the primary system boundary, anticipated or unanticipated transients requiring shutdown of an operating plant, and human errors. (Human errors are discussed in Section V.)

A. Internal Initiators

The main cooling system of a light water reactor (LWR) contains water at high temperature and pressure. If the pressure system boundary should leak or fail, water and steam would be rapidly ejected and a loss-of-coolant accident would be initiated. The outcome of such an event would depend upon the effectiveness of the emergency safety features of the plant in coping with the event.

Another broad class of initiating events is termed transients. These are events that would require plant shutdown, and thus the plant shutdown system must operate effectively and then the decay heat removal system must dissipate the heat generated by the radioactive decay of the fission products. Both classes of events are considered here.

1. The Primary Pressure Boundary

a. General Considerations

The reactor pressure boundaries of LWRs have been a major subject of attention. Most safety-related R&D programs emphasize pressure boundary integrity, possible degradation mechanisms, nondestructive examination (NDE) techniques for inspecting the pressure boundary, and loads, particularly under faulted conditions. Programs on functional performance of pumps and valves under extreme conditions have been relatively limited and deserve more attention. The areas of mechanical components and structural safety have substantial overlaps with regard to seismicity, load combinations, codes and standards, and behavior under transient and accident loads other than seismic; the programs are well conceived and have been meeting their goals.

The NRC fracture mechanics program needs to be examined in the context of national and international work in linear-elastic fracture mechanics (LEFM), elastic-plastic fracture mechanics (EPFM), and general yield fracture mechanics. Taken separately, the NRC program alone is considered inadequate; however, the overall coverage nationally and internationally is excellent. LEFM has been applied directly to nuclear systems for some time. It is recognized that there is substantial overlap among programs on EPFM, but this overlap is considered acceptable in such a relatively new field. The progress in EPFM has been rapid, and this methodology should be ready for application within a year. The ultimate test will be found in application to real systems.

Intergranular stress corrosion cracking (IGSCC) in austenitic stainless steels has been a major problem for BWRs and it has occurred in some PWR piping to a limited degree. Extensive research on this topic has been funded by NRC, EPRI, and industry (e.g., the BWR Owners Group). The phenomenology of IGSCC is now fairly well understood as are materials optimized to eliminate IGSCC in new piping systems. An area that has also been investigated extensively has to do with actions that might be taken to minimize IGSCC in existing systems. These actions include rigorous control of water chemistry, and means to reduce stress levels in sensitized regions (near welds). An example is the application of induction-heating-stress-relief (IHSR) pioneered in Japan, to place the inner surfaces in the heat-affected zone in compression. Despite extensive research and actual plant applications in Japan, there is still some question by the NRC staff regarding the degree of stress reversal attained. Some programs to further quantify the value of the process and to qualify it could be of value.

The steam generator tube degradation/integrity work is a part of a larger overall program funded nationally and internationally. The industry research program has led to a reduction in the rate of tube damage and required tube plugging. Several reactors have units with degraded steam generator tubing. These units will probably continue to operate until the number of tubes plugged is excessive. One area that has not been considered sufficiently using recent accident analysis codes is estimation of the consequences of a transient or some other failure that might lead in turn to the failure of a significant number of tubes. Such failures could lead to the degradation of ECCS function.

- Programs should be funded to apply the best available analytical methods to the assessment of effects of failure of degraded steam generator tubing on reactor transients and accidents.

The final area relevant to the pressure boundary is nondestructive evaluation. The extensive programs need to be examined in the context of other national and international programs. Programs are generally very good, with good rapport among the NDE community. The work seems to be progressing well toward adequate resolution of all issues. Adequate methods are becoming available, but extensive validation campaigns are needed to get code approvals and/or to support NRC acceptance while the code approval process is carried through.

b. Pressure Vessel Rupture

Analyses some years ago by AEC and ACRS resulted in the conclusion that the probability of reactor pressure vessel rupture was acceptably low. Recent reactor occurrences have, however, drawn attention to a possible RPV failure mechanism, not analyzed in the past, which is called cold repressurization.

Significant risk from cold repressurization in reactor pressure vessels could appear near end of life for some vessels. The initiating incident might be a partial blowdown of the PWR secondary side. If there were a flaw in the reactor pressure vessel beltline near the inner surface, if the vessel had undergone severe radiation damage in its beltline, and if the cooling rates were sufficiently rapid to generate high tensile loads around the flaw, fracture might in principle occur. Major programs are planned or are underway by NRC and industry. One area of possible inadequacy is in the ability of NDE to detect flaws near the inner surface. Much of the development of ultrasonic systems optimized for detection in this region has occurred in Europe. The problem is one of quantification of reliability and validation for code approval rather than a feeling that no UT systems exist capable of reliable detection.

- The possibility of cold repressurization of reactor pressure vessels that have undergone large shifts in nil ductility transition temperature provides a potential mechanism for a major failure of the pressure boundary. The substantial programs underway or in the advanced planning stage should be funded and coordinated to assure that the issue is resolved in a 1-3 year time frame.

c. Effects of Water Hammer

An issue of continuing importance is the effect of water hammer and water slugging on PWR and BWR piping systems. Water hammer incidents which have caused damage to piping, valves, valve operators, etc. have been extensively reviewed; however, research aimed at assessment of the severity of water hammer required for damage, and corrective measures, has been quite limited. Little is known concerning the probability of initiating a LOCA by water hammer.

- NRC programs should be established to determine the mode and likelihood of severe damage by water hammer, to permit engineering solution to prevention of damage to the plant and especially safety-related systems.

d. Valves

Valves represent a major possible source for the initiation of accidents, not so much by structural failure, but through loss of function. Definitive studies have been conducted on modes by which valves fail; however, little has been done to develop measures to minimize functional failure. Corrective action will require definitive research programs and close rapport with valve manufacturers. Establishment of this rapport is recognized as very difficult but necessary.

- Industry or DOE programs should be established to develop valves with more reliability of function under normal and abnormal operating conditions.

e. Pipe Design for Seismic Loads

A common complaint from utilities is the adverse effect that conservatisms in seismic design have had on piping systems. Conservatism in damping factors used to calculate response of piping to earthquakes has initiated a chain reaction adversely affecting maintenance, in-service inspection, and general piping reliability. Current regulations result in such a large number of supports and snubbers to comply with postulated seismic loads that it is difficult for maintenance or in-service inspection personnel to obtain access to piping. Furthermore, the rigidity of the systems leaves little margin for errors in installation. The NRC Seismic Safety Margins Research Program (SSMRP) has as one objective the assessment of behavior of piping under seismic loads. Previous programs have confirmed that large damping factors exist under loads, and have also developed inelastic response computer codes. These programs indicate the advantages of basing design on meaningful inelastic response rather than on conservative elastic response. Available records on petrochemical plants confirm that seismic loads rarely, if ever, have failed piping of diameter greater than 4 inches when such systems are left flexible and where little attention has been paid to seismic design. Increased flexibility in nuclear service should result in increased overall safety through better maintenance and in-service inspection and through design to conditions more forgiving under stress.

- A program should be undertaken to examine the relative safety merits of rigid versus flexible seismic design of piping. The program should consider allowance for inelastic behavior of piping and evaluate the potentially adverse impact of an excessive number of hangers and snubbers on in-service inspection and the possibility of crack initiation during normal operation due to snubber malfunction.

f. Decontamination of the Primary System

The primary system of an LWR becomes contaminated in service by the deposition of films of corrosion products consisting predominantly of iron and nickel hydrous oxides, with an appreciable amount of cobalt as the most important radionuclide. During sustained operation of the reactor, a progressive and irreversible increase in the level of radioactivity in the primary piping and equipment occurs, which prevents or inhibits "hands-on" access to these components after the reactor is shut down. Substantial research and development have been undertaken by industry over the years to understand the chemistry and to remove (decontaminate) the accumulated deposits.

An integrated research and development program is needed to establish techniques and procedures for routine on-line decontamination of commercial power reactors.

If such demonstrated techniques could be used industrywide, there would be fewer plant outages, better equipment surveillance and maintenance, enhanced safety of the primary system, and a major and urgently needed reduction in exposure of plant personnel.

- DOE and industry should now proceed with the long-planned, comprehensive program to develop and demonstrate primary system decontamination techniques applicable to LWRs. This program should include techniques which reduce the initial deposition of radioactive materials.

An important large pilot demonstration to decontaminate the primary system of Dresden-1 has been delayed several years by regulatory problems. These should be resolved promptly by NRC. DOE should give high priority to this demonstration and support the follow-on, full-scale decontamination of Quad City-1 using the dilute solution approach to be demonstrated at Dresden-1.

- The proposed decontamination demonstration at Dresden-1 should be undertaken expeditiously.

Because of a difference in reactor water chemistry, a low concentration process is more applicable to BWRs. Significant and encouraging progress has been made at the PNL decontamination test facility which indicates that the advantages of the low concentration process can be extended to all LWRs.

Compatibility of the decontaminating solution with primary system components must be demonstrated, as well as practical and safe methods of disposing of the spent solutions. A large R&D program involving industrial and academic laboratories will be needed.

- Research programs to develop low concentration decontamination processes for both BWRs and PWRs should be continued and expanded.

2. Operating Transients

Plants are typically shut down about five or more times a year, sometimes for planned reasons, but just as often for unexpected reasons. For the most part the unplanned shutdowns are caused by such events as turbine trip and loss of feedwater. From long experience such events have come to be expected, and little further research is needed to understand them. However, two unusual types of transient do warrant further study: station blackout (i.e., loss of all AC or DC power), and events initiated by complicated interactions between the plant control systems.

a. Loss of Electric Power

The possibility of complete loss of electric power supply, either DC or AC, is a significant contributor to total risk for PWRs. In the larger context of protection of the plant itself, the reliability of electric power is vital to all nuclear power plants, because extended failure would lead to severe damage to the core. Improvement of reliability of electric power, either offsite or onsite, would probably be the most effective way to reduce risk to the public and to the industry. It is therefore somewhat surprising that no research to improve the reliability is included in any of the programs the Review Group has considered.

A reasonable first course would be to review carefully the fault trees that analyze electrical failure, to determine which problems are most important and which are most amenable to remedy. A modest program to improve reliability should evolve naturally, and it could be highly cost-effective in reducing risk. Some possible alternatives that might develop are improved diesel generators, alternatives to improved diesels, portable generators with tie-on locations, instrumentation for DC battery banks, automatic testing for weak cells in battery banks, etc.

- A program should be initiated to develop and evaluate methods of providing more reliable electric station power, both AC and DC. Such a program would have significant value and an excellent cost-benefit ratio.

b. Plant Control System Interactions

The traditional approach to LWR design has been to require that reactor protection systems be "safety grade," but to impose no special requirements on other control systems. In recent years, several severe or potentially severe transients in LWRs have been initiated by control system failures. Also, it has been recognized that, in some cases, control system failure might not only cause a challenge to safety systems but also negate or complicate the efficacy of some of the safety systems needed for the transient under consideration.

Research is needed to better define the role of plant control systems in LWR safety so that those changes which are important to safety can be made.

- Research on the role and importance of plant control systems in LWR safety should be greatly augmented.

3. Other Internal Initiators

a. Fires

Since the fire in the Browns Ferry plant, considerable NRC attention has been aimed at reducing the likelihood that fire will cause serious accidents. There seems to be no need for any expanded research in this area.

b. Sabotage by an Insider

The matter of how and when to include design measures and/or other requirements to protect against sabotage by an insider is complex and requires careful, detailed study if a nearly optimal approach is to be developed. The NRC has completed some scoping studies on this matter. The next step in instituting a systematic and sufficiently detailed examination and evaluation of possible approaches to development and adoption of NRC criteria on this matter should be given the necessary priority in the NRC safety research program.

- The NRC should undertake the studies needed to develop design measures and/or other measures which protect against sabotage by an insider while not compromising safety in other respects.

B. <u>External Initiators</u>

1. Earthquakes

Many probabilistic analyses conclude that earthquakes are an important contributor to the overall risk. The validity of this conclusion is hard to assess because of the large uncertainty associated with earthquake analysis. The uncertainty appears both in estimates of the frequency vs magnitude of large earthquakes and in estimating the probability of damage by earthquakes of a given magnitude. Comments on research needed to address these issues are noted in other sections of this report, particularly in the discussion of piping and probabilistic risk analysis. Estimates of the risk from large earthquakes show a great sensitivity to assumptions as to the size of the largest possible earthquakes in a given seismic region.

- The prediction of earthquake magnitude as a function of frequency still has significant uncertainties and is an important area for further research.

2. Floods

Methods for estimating the flood level as a function of flood frequency are still uncertain.

- Research aimed at developing more realistic predictions of flood level as a function of flood frequency is warranted.

3. Wind

The frequency of tornadoes is well known from weather records. The wind loadings and the impact of wind-driven missiles have been well studied. Further research is not needed in this area.

4. External Fires and Explosions

The effects of fires and explosion from sources outside the plant will generally have to be dealt with on the basis of engineering judgment. No research is needed in this area.

II. Response to Accident Initiators

As implied in the previous section, the first level of defense against accidents is to design, build, and operate the plant so as to reduce the frequency of possible accident initiators. The second level of defense is provided by plant systems designed to respond to initiating events and terminate their effects before serious damage occurs. The plant contains a number of systems designed to respond automatically to a variety of initiating events. Most important among these are the reactor shutdown system, the emergency core cooling system, and the heat removal system. On a longer time scale (greater than about 10 minutes) the operator is also required to respond.

A. The Reactor Shutdown System

All reactors are required to have a quick-acting highly reliable system to stop the chain reaction. This is accomplished by rapid (a few seconds) insertion of neutron poisons called control rods into the core. In addition, LWRs have more slowly acting systems which inject boron-containing liquids into the primary coolant. The effect of these poisons on the chain reaction is well understood. The principal question remaining is just how reliably these systems function. Comments on the need for developing better techniques for estimating the failure rate of highly reliable systems are made in Section V.

B. The Emergency Core Cooling System

Following a rupture of the primary system boundary, it would be important to add water to the system to make up for the water lost out of the break. For large breaks, large volumes of water at low pressure would be required, while for small breaks, small volumes of water at high pressure would be required.

1. Large Loss-of-Coolant Accidents

The large loss-of-coolant accident has been studied in great detail for well over a decade. Elaborate thermal-hydraulic codes have been developed and checked against experimental measurements. The most pertinent full system measurements came from the LOFT tests. There is now little doubt that the current NRC methods of determining the effectiveness of the systems for coping with large loss-of-coolant accidents are very conservative.

The LOFT program to study loss-of-coolant accidents (LOCAs) in a small PWR was initiated in 1962. The program was redirected in 1967, and in 1969 LOFT was redesigned to include features that model the emergency core cooling system (ECCS) of a large PWR. Two nuclear-powered tests of large-break loss-of-coolant accidents

had been run by May 1979, showing that the ECCS gave results even more favorable than expected. The facility has since been used to study transients and small-break LOCAs. Thus, the original purpose of the program has been largely fulfilled and the facility has also been useful in working on questions raised by the TMI accident.

The LOFT program has been a success in that it answered satisfactorily the major question of reactor safety for which it was designed, but it does not directly address the key safety problems that are apparent today. The Review Group is reluctant to suggest termination of a facility with such a capable staff and one that represents a large capital investment. But the expense of its operation leads to the conclusion that it is no longer a cost-effective facility. It is unlikely to uncover any major new safety issue. The Review Group has identified a number of high priority areas in which research results are badly needed, and for which the resources saved through phasing out LOFT should be applied.

- The LOFT program should be phased out in an orderly manner. The Re view Group is divided on whether the experimental program for LOFT should be completed in FY-82 or FY-83.

Semiscale is a small mockup of portions of a PWR system. It is versatile and capable of performing experiments on a short-time schedule. It has limitations in size and ability to provide a representative configuration of PWR components, but it is an important facility for carrying out thermal-hydraulics tests. The opinion of industrial representatives is divided on the usefulness of Semiscale, although there is support for a modification that would mock up the once-through steam generator system of Babcock and Wilcox. The cost of operating this system is small enough that we feel its support for the study of thermal-hydraulic phenomena associated with loss-of-coolant accidents should be continued.

2. Small Loss-of-Coolant Accidents

The Reactor Safety Study indicated that the small LOCA is a greater contributor to risk than the large LOCA. The ability to analyze the small LOCA has not kept up with the ability to analyze the large LOCA, in part because of the long computing machine runs required and the resulting buildup of calculational error. In addition, the integral LOCA experiments in the NRC program were designed to test calculations of large LOCAs, and are not well suited to small LOCAs.

The amount of damage to the reactor core from a small LOCA will be very sensitive to the extent and duration of uncovering of the fuel. Some levels of the boiling boundary below the top of the fuel can be tolerated for some time, but the specifics are still more uncertain than is desired. An NRC program to clear up the uncertainty is definitely needed.

- The NRC program on the small loss-of-coolant accident should be continued, to the point where capability to answer important questions of thermal-hydraulics and fuel performance is assured.

C. Decay Heat Removal Systems

Following shutdown of a nuclear plant, the radioactivity in the core continues to generate substantial amounts of heat. To prevent fuel damage, this heat must be removed. The decay heat removal system has this function. Risk analysis indicates that potential failure of this system is an important contributor to the overall risk.

From operational experience, probabilistic risk assessments, and detailed design reviews of specific plants, it is clear that significant improvements in LWR safety could be achieved through improved, more reliable, shutdown heat removal systems. Several countries, including Germany and Switzerland, have required not only highly reliable shutdown heat removal systems but in addition a dedicated bunkered system which provides backup to a loss of all normal offsite and onsite emergency power, and also protection against fire or sabotage. Conceptual design studies of the various principal design alternatives for new and existing LWRs are needed in sufficient detail for development of design and/or performance criteria for improved shutdown heat removal systems.

- A major effort should be undertaken to develop and evaluate improved or alternate approaches to more reliable shutdown heat removal systems, for both the reactor vessel and the containment.

D. Operator Response

The correct operator response to unexpected events depends in large measure on his understanding of the system and on how unambiguously the information presented to him portrays the actual situation in the plant. The accident at TMI revealed problems in both these areas, particularly with regard to understanding unusual transients.

1. Engineering Simulation

The Department of Energy has been instructed by Congress to study the possible usefulness of a national reactor engineering simulator facility. Most technical groups have expressed doubt as to the usefulness of such a facility, a view which this Review Group shares. However, the Review Group does favor a program to develop one or more national system analysis facilities for LWRs. Principal emphasis in the program would be placed on the development of methods to improve present capa-

bilities in system modeling, analysis of severe accidents, and ability to compute in real time or faster. Such simulation facilities would not include a full-scale control room mockup, nor would they be intended for emergency response applications or for operator training. Compromises between physical accuracy and computer running time might be necessary to enable the simulation facility to be used to study large numbers of transients. Such important questions as design alternatives and the effects of operator interaction could be explored parametrically.

Such a system simulation facility(s) should be useful for studies by reactor designers, reactor regulators, and the technical support groups of nuclear utilities. The last group could obtain important generic insights, not now available, from simulators or from actual operational experience. Ultimately, this knowledge base would be reflected in the education and training of reactor operators and supervisors.

- A national LWR system simulation program should be undertaken cooperatively by DOE and NRC. This program would treat both PWRs and BWRs generically. The principal goal should be the development of computational capability to study LWR behavior in real time or faster through a wide range of severe transients including accidents involving extensive core damage.

2. Validated Data

The information presented to the operator should be clear and unambiguous. Considerable work is underway by the industry and the NRC on developing effective means for presenting pertinent information to the operator. These are commented on in Section V. However, experience has shown that an important potential source of error occurs when incorrect information is presented to the operator because of instrument error. Significant industry effort is now addressing this issue. There seems to be no NRC work in this area.

One way in which the reliability of information to the operator can be important to safety is illustrated by the transients involving power supply failure to non-nuclear instrumentation at the Rancho Seco, Oconee, and Crystal River plants. The specific source of difficulty in these cases is being remedied, and current development of parameter display systems may alleviate concerns of this kind.

However, other potential scenarios remain, including those associated with a large earthquake near the plant, for which the reliability of information supplied to the operator might have safety implications. Hence, a reasonably comprehensive examination of this subject should be undertaken.

- A broadened research program should be undertaken on the safety implications of the reliability of information provided to the operator.

E. Rulemaking on Minimum Engineered Safety Features

The NRC is planning a rulemaking hearing on minimum engineered safety features. It is important that a detailed review be carried out to assess the information needs for this hearing, and that any research programs needed to provide additional information on a timely basis be identified and started quickly.

- The NRC should promptly identify those information needs of the rulemaking on minimum engineered safety features of future LWRs that can be supplied in time by safety research, and assign the necessary priority and resources to this task.

III. Fuel Damage and Mitigation of Its Effects

The general topic of damage to nuclear fuel as a result of potential accidents is divided into three parts: 1) the fuel damage process itself, 2) the release of radioactivity from the fuel and its transport to the containment boundary, and 3) the effectiveness of the containment at preventing the release of radioactivity under loads caused by various postulated accidents.

A. The Fuel Damage Process

The topic of fuel damage covers a wide range of possible conditions, from perforation of some of the cladding to total melting. Some minor cladding failure and cracking could result from thermal stresses and other effects of normal operation, including anticipated transients. Serious failures of the cooling system could result in various degrees of overheating that would lead to gross cladding failure and eventually to deterioration and melting of the fuel itself.

The possibility of damage to the fuel is a major safety issue because under normal conditions the UO_2 fuel pellets form an effective trap for most radioactive fission products. Thus any large release of these fission products would be preceded by serious damage to the fuel.

1. Normal Operation and Anticipated Transients

Safety-related research on this item is needed to maintain a low probability of fuel element failure despite plans to increase the burnup routinely achieved with LWR fuel. Low failure frequency limits the amount of radioactive material in the primary system. This facilitates maintenance, repair, and in-service inspection, and therefore helps to keep the integrated occupational man-rem dose at a low value. A lower inventory of radioactivity in the primary system would also reduce any offsite release from certain postulated minor accidents, such as steam generator tube rupture. It would reduce some problems in waste handling.

In addition to an extension of the current research and development on pellet-clad interaction to higher burnups, consideration will have to be given to any possibility of change in the release characteristics of fission gas from the UO_2 fuel pellet during anticipated transients in reactors containing higher burnup fuel. Research in these areas is related to work historically conducted either by industry or under DOE and its predecessors.

- Research on fuel element behavior during normal operation and anticipated transients should be the responsibility of the nuclear industry, or be part of a program of generic studies supported by the Department of Energy (DOE).

2. Super-Prompt-Critical Bursts in Power

This was of considerable interest in the 1960's when reactivity insertion accidents in LWRs appeared to have the potential for bringing a significant amount of oxide fuel beyond the melting point, perhaps to the vaporization point, thereby possibly generating damaging pressure pulses and disruption of the core. However, design changes have made reactivity insertion accidents of this magnitude and speed highly improbable in LWRs, and recent analyses have indicated that for some of the reactivity-insertion accidents still receiving consideration, the power burst would be terminated sooner than was previously calculated.

Hence, it does not appear that additional research in this field is required at this time.

3. Local Melting of Fuel Pins at Power

It has been suggested from time to time that if melting of a significant portion of a subassembly were to occur at high power through flow blockage or some other cause, the event might propagate. Research was deemed necessary to ascertain the reality of such a possibility, and the time scale for detection of the fuel melting and any required mitigative action.

The flow blockage issue is essentially confined to boiling water reactors, whose fuel assemblies have side walls isolating each assembly from its neighbors. The General Electric Company has analyzed such an event, assuming that for some reason one full fuel assembly was blocked, and has concluded that timely detection of the event and shutdown of the reactor would occur, with no significant potential for rapid propagation. The possibility of performing experiments in the Power Burst Facility (PBF) to examine the matter further has been proposed, but, as of now, no such experiments have been scheduled. Additional insight may arise from the experimental program related to degraded and molten cores, which is discussed below.

4. Degraded and Molten Cores

Although recommendations that research was needed on phenomena related to degraded and molten LWR cores were made as many as fifteen years ago, it was the accident at Three Mile Island that provided the impetus for initiation of an NRC research program on this matter. The question is complex, since many different kinds of phenomena are involved, and the situation could in principle be caused by many different initiators, each with its own circumstances.

It will be useful to structure the consideration of degraded LWR cores by classifying conceptual accidents into two categories: (1) accidents that would be terminated before a large part of the core had melted (TMI-2, for example), and (2) accidents that would cause the entire core to melt, threatening the integrity of the reactor vessel and subsequent release of the core to the containment building.

The longer term research programs addressing the first category should be developed in light of two important classes of questions:

(1) Could research significantly influence the course of an accident in which choices available to the operator might affect whether the accident is successfully terminated? If so, how? What research might be done? Is the research feasible technically and financially? How might its usefulness be achieved?

(2) Can research on fuel performance prior to extensive melting lead to improved design of new LWRs or changed designs of existing LWRs? If so, how? How reliable and accurate will estimates of the improvement be?

In the more immediate future, programs may be necessary to ensure to the fullest extent possible a base of information for the degraded core rulemaking hearing.

A similar set of questions must be addressed by that part of the research program that considers phenomena related to molten cores:

(1) Could research on fuel behavior in the reactor vessel during a core melt accident significantly influence the management of an actual accident? If so, how?

(2) Could research on fuel behavior in the reactor vessel during a core melt accident significantly affect the design of current or future LWRs? If so, how?

(3) Are there other objectives of such research than accident management or design changes?

Of course, research programs on degraded cores should also be confined to questions whose answers may have real significance, and which can be well enough defined.

The NRC has announced its intention to hold a rulemaking hearing on "Degraded Core Cooling" to consider how degraded cores should be addressed in the licensing process. To prepare for this hearing, the industry has started the IDCOR program, to carefully review and analyze all available pertinent data. This information and possibly results of new experiments are being used to develop improved analytical tools for degraded core issues. It appears that few, if any, results from currently planned NRC research programs on degraded cores can be expected in time for the

[redacted]

hearing. Specific examples of areas with limited or non-existing data relevant to degraded cores are factors affecting the extent of core melt, gross clad damage, change in fuel geometry, and coolability of degraded cores. Some additional experimental and analytical studies could still be initiated and conducted in time to be useful for the hearing. Consideration should be given to establishing priorities, with this objective as the highest priority. Care should be taken not just to duplicate work of the IDCOR program.

- The NRC should promptly identify that information which it needs for the degraded core cooling rulemaking and which safety research can supply in time for the hearing. It should assign the priority and resources to get the job done, deferring, if necessary, other less urgent degraded core safety research.

5. Three Mile Island Recovery

A careful experimental study of the seriously degraded core in TMI-2 would provide very valuable information on many aspects of core degradation for use in future analysis of such accidents. It is important that enough support be available to obtain the unique information that can be found by detailed examination of this damaged core. It is recognized that the severe institutional and financial problems impeding TMI cleanup must be solved to make this possible. The longer the inspection of this core is delayed, the less trustworthy and the less valuable will be the results of the examination.

- Ample federal funds should be made available for developing timely information through examining the degraded core in TMI-2. The current program of DOE to carry out such investigations should be supported.

6. Power Burst Facility (PBF)

The PBF is a facility unique in this country for the study of fuel failure mechanisms. It provides a service essential to determining damage to an order of magnitude or better as a precursor to inserting more sophisticated experiments into such facilities as NRU or ESSOR. Unless such scoping studies are done beforehand, approval to insert such experiments in other reactors could be difficult or impossible to obtain. For this reason and because of valuable data from the scoping experiments themselves, continued support of PBF is desirable.

On the negative side, some of the recently proposed programs for PBF appear ill conceived and could lead to misinterpretations. This is particularly true of experiments aimed at extrapolation to full core behavior. Such extrapolations could lead to results which are either much too conservative or much too optimistic. The program now proposed should be reviewed for its relevance and modified accordingly.

- Power Burst Facility operation should continue; however, the proposed NRC programs relevant to degraded cores, etc., should be reviewed critically and modified as necessary.

B. Release and Transport of Radioactive Material*

Understanding of the nature of the risks from potential reactor accidents depends heavily upon estimates of the form and amount of radioactivity that might be released from the reactor containment building under a variety of postulated accident conditions. This area can be subdivided into three parts:

(a) The amount and form of radioactivity released from the fuel itself under various conditions of degradation, which will be called the fuel release process.

(b) The amount of removal of radioactivity by plateout, washout, or agglomeration that takes place during transport from the fuel to the containment barrier, which will be called the radioactive transport process.

(c) The modes of containment failure and their effect upon the fraction of the radioactivity that is released to the environment, which will be called the containment failure process.

1. Fuel Release Process

The risk from reactor accidents is thought to be dominated by those failures that would produce serious degradation of the fuel from overheating or melting. Current estimates are based mainly on the Reactor Safety Study (WASH-1400) which contained analyses performed between 1972 and 1975. The WASH-1400 report assumed that in the most serious core melt accidents, well over 50% of the most volatile fission products would be released to the primary system. It also assumed that up to 10% of the less volatile fission products would be released from the fuel. These assumed fractions were based upon laboratory experiments on very small amounts of UO_2. It was noted in the WASH-1400 study that these values were probably conservative. However, a review of the TMI accident and other fuel failures indicates that during such incidents high release fractions of such volatile elements as Cs and I are not only possible but likely. It has been suggested by some, though, that the physical and chemical form of the released radioactivity can have a large impact on the transport processes. In this regard, the question of whether the iodine is released as elemental iodine or as the compound CsI may be of importance. A second question of importance is the size of aerosol particles released and the aerosol density.

* This topic is often referred to as "the source term." In fact, each accident scenario has its unique source term.

2. The Radioactive Material Transport Process

Radioactive gases and aerosols released from the fuel would have to pass through part of the primary system and then into various compartments in the containment structure to pose an increased threat. During this process, there would be potential for plateout on cold surfaces, agglomeration of aerosols into heavier particles which may settle out, or washout by contact with water. For the very worst accidents considered in WASH-1400, it was assumed that a relatively small fraction (less than half) of the radioactivity released from the fuel would be removed by these processes. The method used for estimating the fraction of radioactivity removed by these processes in WASH-1400 may be conservative for many of the accident sequences. This is important because if the estimate of releases of all the major isotopes other than the noble gases were reduced by a factor of 10 or more, the calculated number of acute fatalities would go to zero in every case, and there would be a substantial reduction in other health effects as well. However, for this to change the current view of the nature of reactor accidents, it would have to be established not only that the removal fraction is larger but that it is larger in all the important accident sequences. Research on this transport process should receive a high priority, and should include analysis of transport of fission products in real accidents that have occurred. The following areas are particularly in need of investigation:

(a) Plateout in the primary system.

(b) Washout by sprays or passage through suppression pools and ice condensors.

(c) Rates of agglomeration of very high density aerosols.

(d) Deposition on wet containment surfaces.

(e) Removal during passage through containment fractures.

The NRC has recognized the importance of this issue and recently issued a report entitled "Technical Basis for Estimating Fission Product Behavior During LWR Accidents," NUREG-0772, June 1981, which points out the importance of these removal processes, but concludes that existing knowledge does not warrant major changes in the WASH-1400 values of the removal fractions. In addition, NUS Corporation under contract to DOE is reviewing these issues and suggesting research to resolve them. A report containing their recommendations is to be published shortly. Finally, in preparation for the announced NRC Degraded Core Cooling Hearings, the industry has organized a major review of these issues in their IDCOR program.

The NRC, DOE, and industry have ongoing programs to provide improved understanding of radioactive releases possible after severe reactor accidents. These should encompass a range of conditions and accident scenarios sufficient for broad understanding of the processes of transport and release of the important radioactive isotopes. The NRC program should provide as much information as possible for the degraded core cooling hearings. There is also a need for a long-range program in this area.

C. Containment Effectiveness

As would be expected, accidents with major offsite consequences can occur only if the containment fails. Thus, the causes of possible containment failure and the failure mechanisms themselves are very important to reactor safety. The failure modes could be of three types:

(a) Failure to isolate.

(b) Failure due to excessive internal pressure from static or dynamic loads.

(c) Failure due to melt-through by the molten core.

Failure to isolate refers to failures of the systems that are intended to seal the containment under upset conditions. These failures could result either from equipment (hardware) failures or from human failures either in maintenance or operation of the system.

Overpressure failures could result from a gradual buildup of steam pressure due to a failure of containment heat removal following a serious system failure. They might result from a sudden pressure surge caused by hydrogen detonation or burning, or from a sudden large steam release.

In analyses such as those in WASH-1400, all very large releases are estimated to be the result of catastrophic containment failure either before or shortly after core melt. On the basis of current knowledge it seems clear that if the containment were to remain effective for several hours or more following core melt, substantial reductions in the release fractions would take place. Thus the timing of containment failure would also be important.

1. Containment Isolation

All containments are designed to isolate as a result of signals generated by a wide range of abnormal conditions. For overall safety this isolation system must be highly reliable. Very careful attention to this issue is expected during the design and licensing of the plant. No areas of large uncertainty associated with

this process require specific research programs. However, the general comments on improvements to probabilistic risk analyses apply to analyses done on this system to estimate its reliability.

2. Overpressure Failure

During the past decade, most containment research has been focused on subcompartment dynamic loads for PWRs and dynamic loads in BWR suppression pools as they might arise from steam relief or a LOCA. Most research on the latter has been done by the affected utilities and the reactor vendor, and has progressed reasonably well.

A markedly different kind of containment research has arisen as an aftermath of TMI-2. Partly as a consequence of evaluation of the capability of various types of containment to withstand hydrogen burning, and as a result of the special reviews being given the risk from Zion/Indian Point, the failure mode and pressure are being calculated for a range of types of loading. Failure point estimates are currently being made, on the assumption that the building has been constructed in conformance with the design drawings, although for existing plants test values rather than minimum code values are sometimes used for the strength of steel or concrete.

The Nuclear Regulatory Commission is developing and implementing requirements with regard to hydrogen control capability which frequently take into account the existence and effects of pressures well beyond the normal design point. The rulemaking on degraded cores may lead to a range of new requirements for existing and future reactors. Risk-based analysis is likely to become increasingly important in decision making, and such analyses will surely include the mode, point, and timing of containment failure. A failure of leak tightness seems much more likely than a catastrophic structural failure, but this remains to be assured. It may become important in the estimation of failure probability to allow for the possibility of flaws in fabrication and construction, deterioration due to aging, and design errors. It may become important to determine that pressure below which there is a very high probability of retaining containment integrity.

It appears that it will be necessary to develop a new area of research involving a marriage of the techniques currently used by structural engineers with those employed by reliability engineers. This research will probably be largely analytical since integral experiments are difficult, expensive, and too limited in their range of applicability to be definitive.

- The reactor containment is the ultimate barrier providing protection of the public in the event of a radioactive release from the fuel. The NRC research programs to better understand containment response to loads created by postulated accidents are important and should be supported. A close coupling between structural analysis techniques and the methods of probabilistic risk analysis is needed to make better estimates of the probability of containment failure as a function of increasing internal pressure.

3. Effects of Hydrogen

The possibility of hydrogen generation and release in containments has been a recognized safety issue in water reactors for several decades. Problems with safety are not limited to cases of major fuel failure in a large LOCA but may be associated with several factors, such as rates and mechanisms of formation, concentrations for flammability and explosion, and methods of correction and control.

Questions related to hydrogen were recognized as constituting an early generic safety issue which was considered at one point to be resolved by actions taken in Regulatory Guides.

Recently a generic issue on hydrogen was reestablished and promoted to the limited list of unresolved safety issues. It was one of four issues reported to Congress as new unresolved safety issues in 1981.

This increased concern with hydrogen has resulted from a variety of factors, such as enhanced interest in the effects of degraded cores and the greater implications of hydrogen explosion in smaller containments such as the ice condenser.

The task action plan for attacking the hydrogen problem will be developed this fiscal year, and research programs will be established. This program should be given priority and pursued diligently. Complementary programs by industry should be considered in establishing research scope. The significance of the problem should be established in terms of containment design and volume.

- The NRC is initiating a research program on the issue of evolution and burning of hydrogen in accidents. This program is important and should be pursued expeditiously. Complementary programs by industry should be taken into account in structuring the NRC's research program. The significance of containment design and volume should also be taken into account.

4. Alternative Containment Designs

Alternative containment designs are under consideration that would make the containment more effective at coping with loads imposed by postulated degraded core and fuel melt accidents. The possibility of requiring such measures in future designs will be an important part of the Degraded Core Cooling Rulemaking Hearing. It is essential that prior to the hearing the NRC carry out research on the effectiveness of such measures as containment venting without filtering, containment venting with filtering, and core debris retention in reducing public risk. The effectiveness of such measures compared to other such possible alternatives as the dedicated heat removal system discussed in Section II will be an important issue in the hearing that will require this information.

- The NRC safety research program on mitigation of degraded core and core melt accidents should be modified as necessary to provide the information needed on alternative containment design concepts under consideration, including improved containment cooling, containment venting, venting and filtering, core debris retention, and hydrogen control.

IV. Effects of Potential Accident Releases to the Environment and Their Mitigation

Significant amounts of radioactivity are predicted to be released from the containment in the most serious postulated accidents. It is important to estimate the magnitude and likelihood of the possible consequences of such a release, both as to public health and the damage to property. This information not only provides the basis for risk estimates but is required for establishing the civil protection procedures that might be employed to mitigate the consequences. This section considers the current state of the ability to analyze the consequences of releases of various sizes and types and the impact of proposed mitigating actions.

A. Distribution of Radioactive Materials

Analyses of reactor accidents suggest that two pathways should be considered for the distribution of radioactivity following an assumed accidental release. By far the most important in terms of early health effects and property damage is the release of radioactive gases or fine aerosol particles that would be distributed under the influence of prevailing weather conditions. The second pathway would result from the release of radioactivity into surface or groundwaters.

1. Meteorological Modeling

Most analyses have used a version of a "Gaussian Plume" weather model to calculate the distribution of airborne releases. Many, but not all of these models do not allow for changes in wind direction. The computer code CRACIT developed by Pickard, Lowe, and Garrick includes a detailed treatment of wind changes. All meteorological models used contain numerous simplifying approximations that affect their accuracy; important among these is the treatment of rain. The overall accuracy is probably good for a flat inland site for distances of 5 to 10 miles from the assumed release. However, on sites with special topographical features such as deep river valleys, the local topography introduces features that compromise the accuracy.

The NRC is planning to fund the development of a new improved version of the code CRAC (Consequences of Reactor Accidents Code). The Review Group supports this effort and suggests that particular attention be paid to improving the meteorological modeling. The research should include more detailed weather models than the "Gaussian Plume" model, and particular attention should be given to including improvements in handling effects of rainfall and local topography.

- The methods for calculating the influence of meteorological conditions on the distribution of radioactive material following a postulated accident should be improved. The NRC effort to do this through a new version of CRAC (Consequences of Reactor Accidents Code) should continue to be supported.

2. Hydrological Modeling

In accident scenarios in which it is postulated that the reactor base mat fails following a serious release of radioactive material into the containment, a large amount of radioactive material might enter the groundwater. This released radioactivity could eventually find its way into water supplies and be the source of protracted low-level doses. Because of the dilution by groundwater and the longer time scales for groundwater movement, it is highly unlikely that these doses would contribute to early health effects. However, most risk analyses suggest that such releases might be from ten to a hundred times more likely than a large airborne release. The result is that when the long-term health effects are weighted by their probability, the risk may be of the same order as that from airborne releases. More needs to be known about the distribution of radioactivity by groundwater both for potential reactor accidents and for radioactive waste disposal. It would be reasonable for the NRC to continue work at a modest level in this area.

B. Civil Protection Strategy

All estimates indicate that civil protection actions could reduce significantly the effects of accidental releases of radioactive materials. Considerable regulatory effort has gone into planning of evacuation. Clearly, if people were moved out of the path of the airborne radioactivity they would receive no acute doses. However, if they were in the process of evacuation and failed to get out of the path of the radioactive cloud they would receive higher doses than if they had stayed inside their homes or offices. Thus, for part of the population a strategy of sheltering until the radioactive cloud has passed followed by relocation of those in a contaminated area would be the preferred protective action. The NRC has sponsored work in this general area and the Review Group supports their continuation of this work.

Research in this general area is regarded as important, but the Review Group has not examined the full range of activities by federal, state, and local governments, and therefore makes no recommendations for specific research that may be needed.

C. Biological Effects of Radiation

Since the effects of x rays on radiologists were first recognized, it has been widely realized that radiation in various forms can have serious biological effects. During the last fifty years, extensive research programs have been carried out to determine the quantitative effects on a variety of biological systems including man. Although some aspects of the biological effects of radiation continue to be controversial in scientific circles, these effects are much better understood than those of most of the potentially harmful substances being introduced into the environment by man. The current knowledge is sufficient for reasonable, ealistic, or bounding estimates of biological effects of radiation caused by reactor accidents. The largest uncertainty is in the area of the effects of low levels of radiation.

- The national research programs aimed at a better understanding of the biological effects of radiation should be continued at their present level.

D. Economic Impacts of Large Releases

Estimates of the economic impacts of large accidental releases of radioactivity on property show them to be in the same range as large natural disasters. A major component of these costs is from decontamination of land. Very little research activity exists on procedures for decontaminating large land areas, and a modest research program is warranted.

- A modest research program into techniques for decontaminating large land areas is warranted.

V. Important General Issues

In the review of reactor safety, some technical issues were identified that were so broad and so pervasive that they did not conveniently fall into any one of the preceding categories. These include human error and probabilistic risk analysis (PRA). Because of the special importance of these issues, they are discussed in this separate section.

A. Human Error

Most studies of the likely causes of serious accidents conclude that over 50% of the risk is associated with human failure to perform as intended. This observation includes human errors in design and construction, in maintenance and testing during operation, and, of course, mistakes by operators in response to unusual occurrences. The major goal of research in this area is to reduce the incidence of serious human errors. The current program considers three principal ways of doing this: 1) through better training of personnel, 2) through better system design, 3) through improved procedures. Although some proposed improvements would get broad support, many others may be improvements for some events but could reduce the effectiveness of dealing with other events. Thus it is important to be able to judge the overall impact of any specific change in training, procedures, or design on overall system safety. Because of the unpredictability of human behavior it is often difficult if not impossible to do this by probabilistic methods.

A broad research program is needed to improve understanding of the impact of humans on system reliability, operability, and safety, and of other factors that affect the performance of man-machine systems. While human engineering and the man-machine interface have been subjects of particular interest to NASA, DOD, and other industries and agencies for a number of years, the subjects received little attention from the nuclear industry and the NRC until the TMI accident. Since TMI, both NRC and the industry have launched substantial human factors research programs, but the present knowledge base is inadequate and more research is required.

NRC's program plan for human factors research currently includes studies of human error rate, review of control room design from a human factors perspective, and enhancement of operator selection, training, and performance. The Review Group believes that, in general, the work outlined in this plan is useful and important.

However, the accomplishment of many of the objectives would appear to require more resources than are currently assigned to the work. In many cases, the deliverable results of the research are not well defined. Further, in some areas, particularly the human factors aspects of graphic displays and computer aids to reactor operators, the proposed work appears to be overly duplicative of ongoing or completed industry work. The program associated with LOFT appears to be of marginal relevance for this reason.

It is important that the regulatory standards, criteria, and guidance that the NRC must establish in the human factors area be supported by the best possible knowledge base. It is therefore necessary that NRC identify the information it needs for developing criteria, standards, and regulatory guides and revise its human factors research program as needed to obtain that information. In identifying its information needs and revising its programs, the NRC should be cognizant of past and ongoing research programs of NASA, DOD, the nuclear industry, and other establishments. The revised program should be complementary to and not overly duplicative of industry efforts by owners' groups, INPO, EPRI, and NSAC.

Finally, although the Review Group recognizes that identifying valid ways to measure the effectiveness of human performance will be very difficult, it is a critical element that should be addressed in the research program.

- Probabilistic risk analysis indicates that about half the risk from reactor accidents is attributable to human error. The body of knowledge concerning pertinent human factors is inadequate, and it is important that further research be done in this area to provide an adequate technical basis for regulatory activities. The NRC should establish relative priorities, so that its research program will be structured to obtain the most important information first. The important information is that which can be used to improve control room design, operator aids, and selection and training of operators, which should lead to reduction of rates of human error. Further, the NRC program should coordinated with industry efforts.

B. Probabilistic Risk Analysis (PRA)

Since the accident at Three Mile Island the use of PRA as a tool for helping to understand a variety of safety issues has rapidly increased. This use has ranged from estimating the reliability of specific plant systems such as the auxiliary feedwater and scram systems, to complete integrated estimates of overall plant risks similar to that done in WASH-1400, the Reactor Safety Study. The Review Group supports this use of PRA techniques as one of the tools for reaching a better understanding of plant risks and the reliability of specific systems.

It must be recognized, however, that the techniques are relatively new and therefore there is room for considerable improvement. Three areas needing improvement are: 1) input data on basic failure rates; 2) treatment of common cause failures; 3) analysis of highly reliable systems.

In many cases the failure probabilities required as input data for any quantitative analysis are subject to considerable uncertainty. This can be the result of limited input data or of inapplicability of the general data to the system under analysis. The data on human failure rates comprise just one of a number of cases where considerable uncertainty exists. Significant reseach is needed to develop a more reliable data base.

The term common cause failures refers to two or more failures that occur simultaneously as a result of some common cause. This can be the result of an external event such as a fire, earthquake, etc., or of a common design defect. The normal analysis of a system first looks for single random failures or combinations of single random failures that can lead to system failure. The result of this is called the probability of failure by random events. The analyst must then go back and look for possible dependencies between these previously assumed random failures, and the result is the probability of failure due to common causes. In well-designed systems where the probability of failure by random component failure is calculated to be quite low (less that one in 10,000 per demand), it is customarily found that common cause failures are dominant contributors to system failure. Furthermore, the various sources of these common cause failures are often subtle and hard to identify. Numerous techniques are used to estimate the impact of common causes on system failures, but it is commonly recognized that much more research needs to be done in this difficult area.

The analyses of very well-designed systems sometimes predict failure rates in the range of one in one million (10^{-6}) to one in one billion (10^{-9}) per demand. Although it is possible that such high reliability can be achieved, experience has shown that more often than not such predictions tend to be overly optimistic. Very unlikely events tend to be more frequent than these low probability values suggest. For example, a system out of service for maintenance one minute each year has an unavailability of 2×10^{-6} just for that cause. Natural events that may be

as unlikely as one in a million years could overwhelm probability estimates that otherwise are very low. Another area where probability estimates tend to be highly uncertain is the probability of damage to systems subjected to large ground motions. Much more work is needed in the area of estimating the failure probabilities of very reliable systems.

At a time when the number of analyses being requested by the NRC is increasing rapidly, it becomes very important to standardize as much as is practical the techniques to be used. If done with care, this should improve the quality and reproducibility of the results.

The NRC currently has research programs addressing each of these issues and because of the importance of PRA techniques these programs should be expanded.

- The use of probabilistic analysis is rapidly expanding. A significantly larger research program is needed to improve and standardize the methodology as much as possible. Particular areas that need improvement are: methods of handling common cause failures, data collection and analysis, accuracy of the estimated frequency of very low probability events, and assurance of the quality of analyses.

VI. Organizational and Institutional Comments

A. NRC Safety Research

The NRC's safety research program evolved to its present form over an interval of about eight years. This form is now responsive to needs as they developed over that period. The program is not logically or optimally structured in coverage and emphasis to respond to questions of currently perceived urgency. The importance of the loss-of-coolant accident in the historical development of the NRC's regulatory methods has led to focus of research on this topic, with most of this work devoted to the large LOCA. It seems appropriate at this point to change the emphasis of the program. Some of this change is underway, and suggestions are made throughout this report for additional changes.

The program heavily emphasizes reduction or prevention of consequences once an accident is assumed to have started, since the regulatory staff has itself adopted this primary objective. As said below, this emphasis seems appropriate as a guideline, but it should not be rigid. On numerous occasions, NRC research has been necessary to improve reactor safety, and without doubt more such instances will arise.

Most of the NRC's programs are strong, and together they constitute one of the more important research programs in the country, in size, importance, and product. The program is broadly accepted by the technical community, especially where it is more fully understood and its impact is greatest. The program has substantially improved the understanding of many of the reactor safety issues it has explored, and it has greatly improved the underpinning of NRC's regulatory actions on light water reactors.

The Review Group found a widely held perception among reactor safety researchers that the results of research do not lead to corresponding changes in regulations, particularly in those cases in which the results showed the regulations to be based on conservative assumptions. Even after discounting the tendency of researchers to feel their results are not used enough, this criticism appears to be essentially valid. Three examples related to Appendix K to 10CFR50, which resulted from the ECCS Rulemaking hearings in 1972-73, stand out. Appendix K has not been altered in spite of a considerable body of experimental data from the NRC's own research programs showing several of the prescribed features to be very conservative. Chief among those is perhaps the decay heat formulation commonly referred to as ANS + 20%. This formula, to which the analysis is sensitive, is high by a

good 25%. Similarly, data obtained since Appendix K was issued show that the zirconium oxidation rate in steam is less than predicted by the prescribed formulation. Finally, the integral experiments conducted on LOFT indicate that the clad temperature rise in a large-break LOCA will be hundreds of degrees below the value computed by the Appendix K methodology.

Although it is probably true that the NRC staff frequently incorporates new data into their engineering judgments, more formal incorporation of new and significant results is desirable. It is true that the NRC frequently faces industry (vendor or utility) opposition, active or passive, in making regulatory changes even when the proposed change would relax the rules. The industry tends to grow "comfortable," or it learns to live with a particular regulation. The prospect of the procedures involved in a change gives more concern than is compensated by the prospect of a relaxed specification. There is a tendency to retain the outdated assumptions in regulatory actions to provide extra overall safety margin, even when the conservatism has become assured.

The NRC program is in many areas weak in timely production of results, partly because of inflexibility and partly because the requirements for new reseach were not recognized sooner. The inflexibility has several origins, which are addressed in a recommendation below.

A basic criticism heard many times was that the Commission fails to set and communicate policy direction to the staff. There is even a lack of general instructions to the research staff as to what areas of research are considered important. In the absence of direction, the staff does not have an adequate basis to set research priorities and establish schedules which conform to the responsibilities faced by the Commission. This has been one of the causes of the uncertainty in goals that has been observed. The Review Group did not make any in-depth study of this organization problem. However, various individual members found such criticism to conform to their own observations. It goes without saying that an adequate solution to this problem is vital to an adequate reactor safety program, both inside and outside the NRC.

The planning and management of research have probably been diluted by the recent reorganization that combined the Office of Nuclear Regulatory Research and the Office of Standards, On the other hand, the Office of Nuclear Regulatory Research has never become well integrated with the licensing staff, and the new reorganization may ease this problem.

- The Commission should encourage and promote the visible integration of the results of safety research into the regulatory process. Regulatory requirements should not rest on bad science when good science has become available. If retention of conservatism is desired because of uncertainties or for other reasons, this should be done through the application of explicit safety factors to a calculation based on best available methods, in accordance with good engineering practice.

- The limitation of NRC's programs to "confirmatory research" should be removed, so that exploratory research and research to improve safety can be undertaken when this looks like the better course to follow.

- It is reasonable that the prevention of accidents should be the principal area for industrial safety research, and prevention or limitation of public consequences the principal area for NRC. This difference should be retained as a guideline but not as an absolute boundary between the programs.

- The flexibility of the NRC's research program should be improved. An easier way to transfer funds from one decision unit to another should be established. The methods of inter-office coordination and concurrence on new programs should be simplified to prevent a single individual or Office from being able to block programs that have broad support. Subject only to Commission agreement, the Office of Research should be permitted to undertake research that is expected to be important in the long run, even if there is no immediate user need. Of course, this should not displace research for which a user need has been defined and agreed on.

- NRC's long-range research plan (NUREG-0740) should be restructured to follow lines of Agency objectives. This should infuse it with a more up-to-date logical structure and discourage the tendency to support doing more of what has already been done. Introduction and continuation of guidance at the Commission level will be of fundamental importance in confirming the objectives and restructuring the plan. The restructured long-range plan should be more specific in defining the deliverables in each program.

B. The Role of DOE in Light Water Reactor Safety Research

Public Law 96-567 (Nuclear Safety Research, Development, and Demonstration Act of 1980) states that: "a proper role of the Federal Government in assuring nuclear power plant safety, in addition to its regulatory function, is the conduct of a research, development, and demonstration program to provide important scientific and technical information which can contribute to sound design and safe operation of these plants." It is the intent of the law that this program be carried out by the Department of Energy.

The DOE maintains a number of national laboratories which are a reservoir of talent and facilities for conducting reactor safety research and design studies. There is little doubt that a valuable program can be formulated to make effective use of these resources. PL 96-567 specifies a number of areas in which the DOE should establish research programs. Although the Review Group does not believe that DOE is necessarily the best agency to conduct each of the types of research specified, there will probably be enough flexibility to permit DOE to formulate a sound program. However, DOE does not now have the staff structure necessary for planning and conducting this program. In response to the passage of PL 96-567, the Office of Nuclear Power Systems of DOE has conducted a study of the kind of program it would propose and has prepared a draft of the report to Congress. The Review Group found the first draft of this report to contain too many generalities to give much confidence that a well-focused research program will be formulated.

One section of PL 96-567 calls for a study of the need for a "reactor engineering simulator facility." The DOE appears to be reaching a negative conclusion regarding such a need, a finding in which the Review Group concurs. The Review Group does believe, however, that there is need for more detailed and more physically accurate mathematical models of nuclear power plants and rapid calculational methods that incorporate these models, a need discussed elsewhere in this report.

One research project which should especially receive DOE support is the study of the damaged core in the Three Mile Island, Unit 2 plant. DOE has already initiated such a project. This core is a national resource in the sense that it contains more information about the progression of fuel damage during an accident and the characteristics of degraded cores than could be obtained practically in a costly planned research program. The DOE program to study the TMI-2 core is supported in a recommendation elsewhere in this report.

Other areas of safety research which the Review Group finds somewhat more appropriate to DOE than NRC include improved steam generator design, increased reliability of nuclear plant AC and DC electrical systems, and development of more reliable valves.

- DOE should form a strong staff of technical and managerial personnel knowledgeable and experienced in the subject of water reactor safety, to develop and implement programs of safety research in this field.

- DOE should develop a program of generic research to improve water reactor safety, and assume a substantial responsibility in the area of accident prevention to supplement other programs designed to reduce the likelihood of nuclear power plant accidents.

C. Research in Other Countries

Research on safety of nuclear power plants is extensive in some other countries, especially in Japan and West Germany. These programs provide an effective supplement to American research. The participation of the NRC and industry in foreign programs (e.g., Halden, Marviken, LOCA programs) has been beneficial, as has foreign participation in U.S. programs (e.g., LOFT, PBF). Interaction with research programs in other countries has been effective in preventing unnecessary duplication of work. DOE on its entry into water reactor safety research should also begin to take advantage of research done abroad.

- DOE should assume a more active role in the international aspects of water reactor safety research.

D. Freedom of Publication

Publication and communication of results of water reactor safety research programs by NRC, EPRI, and foreign countries have generally been good. A steady stream of reports is issued in this field, compared to a bare trickle about ten years ago. However, proprietary considerations severely limit the dissemination of results of research by reactor vendors and utility owners' groups.

In recent years, the publication and distribution of results of DOE's research on reactor-related subjects have been restricted. This policy, which was adopted in order that results of the research could be used to trade for results of related research by other countries, is mostly applied in the fast breeder field. Research on water reactor safety should be treated differently in this respect from developmental reactor research. Freedom of flow of information should be the rule, in recognition of the fact that reactor safety everywhere benefits all.

- DOE's water reactor safety research program should follow a policy of free and open publication and distribution of reports. However, this policy should permit protection of proprietary information of industry participants.

E. The Role of Risk Assessment

Probabilistic risk assessment can in principle be used to rank programs on a cost-benefit basis. It is important that this practice be started. However, other factors must also be used in arriving at such a ranking. Among these are timing of the program compared with timing of research needs, the probability of success, the impact the research may have, and the extent of resources required.

- A policy should be instituted whereby probabilistic risk assessment is used as one tool in establishing priorities of research programs.

END

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